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APPLIED HEALTH PHYSICS AND SAFETY

ANNUAL REPORT FOR 1969

OAK RIDGE NATIONAL LABORATORY

operated by

UNION CARBIDE CORPORATION

for the

U.S. ATOMIC ENERGY COMMISSION

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HEALTH PHYSICS DIVISION

APPLIED HEALTH PHYSICS AND SAFETY ANNUAL REPORT FOR 1969

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AUGUST 1970

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
U. S. ATOMIC ENERGY COMMISSION

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2.0 CONTRIBUTIONS

The data for this report were contributed by: H. H. Abee, Environs Radiation Monitoring Section; R. L. Clark, Radiation and Safety Surveys Section; E. D. Gupton, Applied Radiation Dosimetry Section; A. D. Warden, Applied Health Physics and Safety Associate Department Head; D. C. Gary, Industrial Safety Engineering.

3.0 VISITORS AND TRAINING GROUPS

During 1969 there were 41 visitors to Applied Health Physics and Safety, as individuals or in groups, for training purposes. Table 12.1 is a listing of training groups which consisted of four or more persons.

Table 3.1 Training Groups in Applied Health Physics and Safety Facilities during 1969

Facility	Number	Training Period
University of Arkansas (Radiological Health)	6	3/25/69 - 3/28/69
TVA - Chattanooga, Tennessee	5	4/8/69 - 4/8/69
ORAU Ten-Week H.P. Course	4	5/27/69 - 6/4/69
Tenn. Tech. University (Sanitary Engineering)	6	4/21/69 - 4/21/69
AEC Fellowship	7	6/23/69 - 8/29/69
University of Tenn. Co-ops* (Physics Department)	8	1/1/69 - 12/29/69

*In four separate groups of one academic quarter.

4.0 SUMMARY

There were no accidental releases of gaseous or liquid waste from the Laboratory which were of a reportable level as defined in AEC Manual Chapter 0524. The concentration of radioactive materials in the environs was well below the maximum levels recommended by the ICRP, FRC, and AEC.

No employee received an external or internal radiation dose of a level that required a report to the AEC. The highest whole body dose received by an employee was about 3.8 rem or 32 percent of the maximum permissible annual dose. There were no cases of internal exposure where the deposition of radioactive materials within the body was estimated to have averaged greater than one-half of a maximum permissible body burden.

There were 12 unusual occurrences recorded during 1969, which is the lowest number recorded since the present system of reporting unusual occurrences was established in 1960. The second lowest number for any one year was 16, the number reported for 1967. The average number reported for the past five years was 22.

The ORNL safety record for 1969 was the second best in the history of the Laboratory. There were only two Disabling Injuries reported during the year, and there was a decrease in number of both Serious Injuries and Medical Treatment Cases as compared with 1968. The Disabling Injury Frequency Rate for 1969 was 0.27, as compared with the average rate of 0.94 for the previous five years, 1964–1968.

5.0 ENVIRONS MONITORING

The Health Physics Division monitors for airborne radioactivity in the East Tennessee area by the use of three separate monitoring networks. The local air monitoring (LAM) network consists of 22 stations which are positioned in relation to ORNL operational activities; the perimeter air monitoring (PAM) network consists of nine stations which are located on the perimeter of the AEC controlled area; and the remote air monitoring (RAM) network consists of eight stations which are located outside the AEC controlled area at distances of from 12 to 75 miles from ORNL.[1] The monitoring networks provide for the collection of (1) airborne radioactivity by air filtration techniques, (2) radioparticulate fallout material by impingement on gummed paper trays, and (3) rainwater for measurement of fallout occurring as rainout. The filter data are representative of radioparticulate matter which might be considered respirable; the gummed paper data are representative of radioparticulate fallout; and the rainwater data provide information on the soluble and insoluble fractions of the radioactive content of fallout material.

Low-level radioactive liquid waste originating from ORNL operations are discharged, after preliminary treatment, to White Oak Creek, which is a small tributary of the Clinch River. Liquid waste releases are controlled so that the resulting average radioactive concentrations in the Clinch River are well below the maximum permissible concentrations for waste released to uncontrolled areas as recommended in Annex 1, Table II, of AEC Manual Chapter 0524.

The radioactive content of the White Oak Creek discharge is determined at White Oak Dam which is the last control point along the stream prior to entry of White Oak Creek waters into Clinch River waters. Water samples are collected also at a number of locations along the Clinch River, beginning at a point above the entry of waste into the River via White Oak Creek and ending at Center's Ferry (near Kingston, Tennessee) about 16 miles downstream from the confluence of White Oak Creek and the Clinch River. Water samples are analyzed for gross radioactivity and for specific radionuclides present in detectable quantities. The concentration of each nuclide detected is compared with its respective MPC_w value as specified by AEC Manual Chapter 0524, and the resulting fractions summed to arrive at the percent $(MPC)_w$ in the Clinch River.

Samples of ORNL potable water are collected daily, composited and stored. At the end of each quarter these composites are analyzed radiochemically for ^{90}Sr content and are assayed for long-lived gamma emitting radionuclides by gamma spectrometry.

Raw milk samples are collected at 12 sampling stations located within a radius of 50 miles from ORNL. Samples are taken on a weekly basis from eight stations which

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[1] For maps showing location of stations, see ORNL-4423, Applied Health Physics and Safety Annual Report for 1968.

are located outside the AEC controlled area within a 12-mile radius of ORNL. Samples are collected every five weeks from the four remaining stations, all of which are located outside the 12-mile radius up to distances of about 50 miles. The purpose of the milk sampling program is twofold: first, samples collected in the immediate vicinity of ORNL provide data by which one may evaluate the possible effect of waste releases originating from ORNL operations; second, samples collected remotely to the immediate vicinity of the ORNL area provide background data which are essential in establishing a proper index from which releases of radioactive materials originating from Oak Ridge operations may be evaluated.

Background gamma radiation measurements are made monthly at a number of locations throughout other portions of the East Tennessee area. These measurements are taken with calibrated GM and scintillation type detectors at a distance of three feet above the surface of the ground.

River bottom sediments in the Clinch and Tennessee Rivers have been surveyed and analyzed annually since the year 1951 for the purpose of providing data relative to the dispersion of radioactive waste released from Oak Ridge operations to the Clinch River. However, in keeping with ORNL policy to reduce operating costs during fiscal year 1969 and as the amount of radioactive materials released to the River during calendar years 1967 and 1968 was less than any previous year, no river survey was performed during the summer of 1969.

Fish from the Clinch River are sampled during the spring and summer each year and analyzed for their radioactive content. The radionuclide concentration in fish are related quantitatively to potential human intake of radioactivity through consumption of fish.

5.1 Atmospheric Monitoring

5.1.1 Air Concentrations - The average concentrations of radioactive materials in the atmosphere, as measured by filtration methods provided by the LAM, PAM, and RAM networks during 1969, were as follows:

Network	Concentration (μCi/cc)
LAM	0.26×10^{-12}
PAM	0.16×10^{-12}
RAM	0.16×10^{-12}

The LAM network value of 0.26×10^{-12} μCi/cc is about 0.01 percent of the $(MPCU)_a$[2] based on occupational exposure of 3×10^{-9} μCi/cc. Both the PAM and RAM network values represent ~0.2 percent of the $(MPCU)_a$ of 1×10^{-10} μCi/cc applicable to waste released to uncontrolled areas. A tabulation of data for each station in each network is given in Table 5.1. The weekly values for each network are illustrated in Table 5.2.

The number of radioactive particles collected on the air monitoring filters is shown in Table 5.1. Data are given on both the activity range of the particles and to the total number of particles per 1000 cu. ft. of air sampled.

5.1.2 Fallout (Gummed Paper Technique) - Radioparticulate fallout as measured by the LAM network decreased by a factor of 1.8 from the value measured in 1968. The values measured by the PAM and RAM networks decreased by factors of 19 and 38, respectively, from the 1968 values. Table 5.3 gives the network averages by weeks. Table 5.4 gives a tabulation of data for each station within each network.

5.1.3 Atmospheric Radioiodine (Charcoal Collector Techniques) - Atmospheric radioiodine measured by the perimeter stations averaged 0.018×10^{-12} μCi/cc during 1969. This is only about 0.02 percent of the maximum permissible inhalation concentration of 1.0×10^{-10} μCi/cc applicable to waste released to uncontrolled areas. The maximum value observed at any one station for one week was 0.49×10^{-12} μCi/cc. This value was measured at PAM 31, the perimeter station located at Kerr Hollow. Table 5.5 compares the weekly discharge of radioiodine from ORNL stacks[3] with the average concentration of radioiodine measured by the perimeter stations.

The average radioiodine concentration measured by the local stations was 0.31×10^{-12} μCi/cc. This is about 0.01 percent of the maximum permissible concentration for occupational exposure. The maximum value observed on any one station for one week was 5.3×10^{-12} μCi/cc. This value was observed at LAM 6 (near the Special Materials Shop). Table 5.6 gives ^{131}I data for both the Plant area (LAM's) and the perimeter area monitors.

5.2 Water Analyses

5.2.1 Rainwater - The average concentration of radioactivity in rainwater collected from the three networks during 1969 were as follows:

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[2]The $(MPCU)_a$ is defined as the maximum permissible concentration for an unknown mixture of radioisotopes in air. AEC Manual Chapter 0524, Appendix, Annex 1, gives exposure values applicable to various mixtures of radionuclides and establishes guide lines for deriving the $(MPCU)_a$.

[3]"Summary of Waste Discharges", Weekly Reports, 1969, L. C. Lasher.

Network	Concentration (μCi/cc)
LAM	0.35×10^{-7}
PAM	0.34×10^{-7}
RAM	0.47×10^{-7}

The value observed on the PAM network was the same as that measured during 1968. The average values for the LAM and RAM networks are not significantly different from the average value for the PAM network. The average values for each station are shown in Table 5.7; the average values for each network for each week are given in Table 5.8.

5.2.2 Clinch River Water - A total of 12,247 curies of tritium and 13 beta curies of radioactivity other than tritium were released to the Clinch River during 1969 as compared to 9,685 curies of tritium and 16 beta curies of other radionuclides released in 1968 (Table 5.9). Yearly discharges of specific radionuclides to the Clinch River, 1965 through 1969, are shown in Table 5.10.

The calculated average concentrations of the significant radionuclides in the Clinch River at Clinch River Mile (CRM) 20.8 (the point of entry of White Oak Creek into the River) are presented in Table 5.11. The concentration of tritium in the River represented 0.11 percent of the $(MPC)_w$ for tritium in uncontrolled areas and the concentrations of other radionuclides when compared to their respective $(MPC)_w$ values represented 0.36 percent of the $(MPC)_w$ for those radionuclides in uncontrolled areas. The percent $(MPC)_w$ did not exceed 1.2 percent for any month during 1969 (Table 5.12).

The measured average concentrations of radionuclides other than tritium in Clinch River water at CRM 23.1 (above the entry of White Oak Creek) was 0.30 percent of the $(MPC)_w$ (Table 5.11). A significant concentration of ^{60}Co was detected in the Clinch River at CRM 23.1 during the first quarter of 1969. The contamination came from a source other than Laboratory operations (see Section 5.2.3 Potable Water).

The measured concentrations of radioactive materials other than tritium at CRM 4.5 (near Kingston, Tennessee) represented 0.41 percent of the $(MPC)_w$ values applicable to wastes released to uncontrolled areas.

5.2.3 Potable Water - The average concentrations of ^{90}Sr in potable water at ORNL during 1969 were as follows:

Quarter Number	Concentration ^{90}Sr (μCi/ml)
1	0.68×10^{-9}
2	0.23×10^{-9}
3	0.36×10^{-9}
4	0.45×10^{-9}
Average for Year	0.43×10^{-9}

The average value of 0.43×10^{-9} represents 0.14 percent of the $(MPC)_w$ for drinking water applicable to individuals in the general population.

Cobalt-60 was detected in the potable water at ORNL during the first quarter of 1969. The composite sample for this period showed a ^{60}Co content of 0.03 d/m/ml. This value is ~ 0.04 percent of $(MPC)_w$ for soluble ^{60}Co for application to uncontrolled areas.

The source of this contamination was found to be a contaminated branch (Braden Branch) that flows into Melton Hill Lake at about CRM 50.5 upstream from the intake of the Oak Ridge City (ORNL potable water source) water supply. The ^{60}Co content of the branch was (when sampled in February, 1969) 67 d/m/ml or ~ 60 percent of the $(MPC)_w$ for ^{60}Co. Subsequent investigation revealed that the contamination resulted from a leaking radioactive waste storage tank belonging to a private nuclear industry located near Braden Branch.

5.3 Milk Analyses

The average concentration of ^{90}Sr in raw milk samples collected within a 12-mile radius of the Laboratory during 1969 was 17.8 pCi/l. The average concentration of ^{90}Sr in samples collected between 12 miles and 50 miles from the Laboratory was 15.7 pCi/l. These results would indicate that the ^{90}Sr content of milk in the Oak Ridge area is from sources other than the Laboratory. Table 5.13 presents the weekly average concentration of ^{90}Sr in raw milk collected from the immediate environs of Oak Ridge.

The average concentration of ^{131}I in raw milk samples collected within a 12-mile radius of the Laboratory, as well as the samples collected between 12 miles and 50 miles from the Laboratory, was below the minimum detectable level of 10 pCi/l, except for weeks 20, 22, 31, and 41. The highest average for any one week, the eighteenth week, was 11.1 pCi/l. (When levels are below 10 pCi/l, for averaging purposes a value of 5 pCi/l is assumed.) Table 5.5 includes the weekly average concentration of ^{131}I in raw milk collected at the stations within a 12-mile radius of the Laboratory and the weekly discharge of ^{131}I from the ORNL stacks.

The average yearly values for both ^{90}Sr and ^{131}I fall within the limits of FRC Range I daily intake guides, if one assumes an intake of 1 liter of milk per day.

5.4 Background Measurements

Background measurements were taken at a number of locations (established in 1961) in the East Tennessee area during routine servicing visits to the remote air monitoring stations. Measurements were made at each location on a frequency of once each five weeks. The average background level during 1969 as measured at these stations was 0.012 mR/hr. Average background readings and the location of each station are presented in Table 5.14.

Background measurements made monthly with a calibrated GM monitor at five selected locations adjacent to the ORNL area yielded an average background reading of 0.013 mR/hr during 1969. Corresponding measurements made at 53 locations on the ORNL site gave an average background of 0.084 mR/hr. The average background level measured in the Oak Ridge area in 1943 prior to the start-up of the Oak Ridge Graphite Reactor was 0.012 mR/hr.

5.5 Annual Survey of the Clinch and Tennessee Rivers

The annual survey of the Clinch and Tennessee Rivers was performed by the Applied Health Physics and Safety Section during the summer of 1969. The 1969 survey extended downstream through Guntersville Reservoir and was somewhat more extensive than either the 1966 or 1967 surveys. The techniques and procedures used are described in ORNL-2847, Radioactivity in Silt of Clinch and Tennessee Rivers.

The 1969 survey showed the dispersal pattern of radioactive silt in the Clinch River below the outfall of White Oak Creek to be essentially the same as in 1967 and the average levels of activity measured to be slightly lower than in 1967 except in Melton Hill Reservoir. Gamma activity in the bottom sediments of Melton Hill Reservoir were found to be a factor of 2-3 times higher than that measured in 1967. These higher levels resulted from a release of ^{60}Co into the Reservoir via Braden Branch in late 1968 (see Section 5.2.3 Potable Water). The gamma count data indicate that the suspended solid portion of the release was deposited primarily in the sediment of Melton Hill Reservoir.

Table 5.1 Concentration of Radioactive Materials in Air—1969
(Filter Paper Data—Weekly Average)

Station Number	Location	Long-Lived Activity 10^{-13} μCi/cc	No. of Particles by Activity Ranges[a]					Particles Per 1000 ft^3
			10^4–10^5 d/24 hr	10^5–10^6 d/24 hr	10^6–10^7 d/24 hr	> 10^7 d/24 hr	Total	
			Laboratory Area					
HP-1	S 3587	2.0	0.00	0.00	0.00	0.00	0.00	0.00
HP-2	NE 3025	2.9	0.04	0.00	0.00	0.00	0.04	< 0.01
HP-3	SW 1000	2.4	0.00	0.00	0.00	0.00	0.00	0.00
HP-4	W Settling Basin	2.5	0.00	0.00	0.00	0.00	0.00	0.00
HP-5	E 2506	2.8	0.06	0.00	0.00	0.00	0.06	< 0.01
HP-6	SW 3027	3.0	0.10	0.00	0.00	0.00	0.10	< 0.01
HP-7	W 7001	2.5	0.00	0.00	0.00	0.00	0.00	0.00
HP-8	Rock Quarry	2.5	0.00	0.00	0.00	0.00	0.00	0.00
HP-9	N Bethel Valley Rd.	2.5	0.00	0.00	0.00	0.00	0.00	0.00
HP-10	W 2075	2.6	0.02	0.00	0.00	0.00	0.02	< 0.01
HP-16	E 4500	2.5	0.00	0.00	0.00	0.00	0.00	0.00
HP-20	HFIR	2.7	0.00	0.00	0.00	0.00	0.00	0.00
Average		2.6	0.02	0.00	0.00	0.00	0.02	< 0.01
			Perimeter Area					
HP-31	Kerr Hollow Gate	1.5	0.00	0.00	0.00	0.00	0.00	0.00
HP-32	Midway Gate	1.8	0.00	0.00	0.00	0.00	0.00	0.00
HP-33	Gallaher Gate	1.2	0.00	0.00	0.00	0.00	0.00	0.00
HP-34	White Oak Dam	1.5	0.04	0.00	0.00	0.00	0.04	< 0.01
HP-35	Blair Gate	1.6	0.00	0.00	0.00	0.00	0.00	0.00
HP-36	Turnpike Gate	1.8	0.02	0.02	0.00	0.00	0.04	< 0.01
HP-37	Hickory Creek Bend	1.5	0.00	0.00	0.00	0.00	0.00	0.00
HP-38	E EGCR	1.6	0.02	0.00	0.00	0.00	0.02	< 0.01
HP-39	Townsite	1.8	0.04	0.00	0.00	0.00	0.04	< 0.01
Average		1.6	0.01	< 0.01	0.00	0.00	0.01	< 0.01
			Remote Area					
HP-51	Norris Dam	1.6	0.04	0.00	0.00	0.00	0.04	< 0.01
HP-52	Loudoun Dam	1.7	0.00	0.00	0.00	0.00	0.00	0.00
HP-53	Douglas Dam	1.6	0.02	0.00	0.00	0.00	0.00	< 0.01
HP-54	Cherokee Dam	1.5	0.00	0.00	0.00	0.00	0.00	0.00
HP-55	Watts Bar Dam	1.2	0.02	0.00	0.00	0.00	0.00	< 0.01
HP-56	Great Falls Dam	1.7	0.06	0.00	0.00	0.00	0.06	< 0.01
HP-57	Dale Hollow Dam	1.6	0.02	0.00	0.00	0.00	0.02	< 0.01
HP-58	Knoxville	1.8	0.00	0.00	0.00	0.00	0.00	0.00
Average		1.6	0.02	0.00	0.00	0.00	0.02	< 0.01

[a]Detection limit – 10^4 d/24 hrs per particle.

Table 5.2

Concentration of Radioactive Materials in Air
As Determined from Filter Paper Data - 1969
(System Average - by Weeks)

Week Number	Units of 10^{-13} μCi/cc			Week Number	Units of 10^{-13} μCi/cc		
	LAM's	PAM's	RAM's		LAM's	PAM's	RAM's
1	1.0	0.49	0.61	29	4.5	2.9	2.6
2	1.3	0.78	0.80	30	2.8	2.0	1.9
3	1.1	0.71	0.73	31	6.6	3.9	4.2
4	0.9	0.36	0.45	32	6.8	4.4	4.1
5	1.2	0.49	0.42	33	3.5	1.9	2.2
6	0.7	0.52	0.48	34	3.5	2.0	2.2
7	1.6	0.73	0.70	35	3.8	2.8	2.7
8	1.2	0.71	0.72	36	1.2	1.1	0.84
9	0.8	0.53	0.56	37	3.2	2.1	2.2
10	1.2	0.92	0.88	38	1.9	0.83	1.1
11	2.2	1.3	1.1	39	1.4	0.82	0.73
12	2.2	1.1	1.2	40	2.1	1.3	1.4
13	0.8	0.84	0.78	41	1.9	1.3	1.6
14	2.4	1.2	1.3	42	2.2	1.4	1.3
15	2.6	1.5	1.8	43	2.4	1.4	1.5
16	2.3	1.1	1.3	44	1.1	0.79	0.87
17	2.7	1.2	1.4	45	0.8	0.56	0.54
18	2.6	1.6	1.3	46	1.4	0.78	0.78
19	3.8	2.5	2.2	47	0.93	0.73	0.86
20	4.5	2.6	2.7	48	1.20	0.79	0.80
21	4.2	2.7	2.7	49	1.0	0.68	0.67
22	6.6	3.9	4.1	50	1.4	0.49	0.52
23	5.2	3.7	3.5	51	1.2	0.71	0.48
24	4.9	2.8	2.8	52	0.69	0.39	0.45
25	6.0	4.3	3.6				
26	3.9	2.7	2.6	Average			
27	4.2	3.2	3.1	1969	2.6	1.6	1.6
28	4.0	2.8	2.2	1968	2.9	1.6	1.6

Table 5.3

Radioparticulate Fallout Measurements[a]
As Determined by Autoradiographic Techniques - 1969
(Gummed Paper Data - System Average by Weeks)

Week Number	Particles/ft^2			Week Number	Particles/ft^2		
	LAM's	PAM's	RAM's		LAM's	PAM's	RAM's
1	0.50	0.00	0.00	29	1.25	0.00	0.00
2	0.00	0.00	0.00	30	0.00	0.00	0.00
3	0.25	0.00	0.00	31	1.00	0.00	0.00
4	0.08	0.00	0.13	32	0.73	0.00	0.00
5	0.17	0.22	0.00	33	0.33	0.00	0.00
6	0.00	0.00	0.00	34	0.08	0.00	0.00
7	0.17	0.00	0.00	35	2.08	0.00	0.00
8	0.42	0.00	0.00	36	0.33	0.00	0.00
9	0.08	0.00	0.00	37	1.25	0.00	0.00
10	0.00	0.11	0.00	38	0.08	0.00	0.00
11	0.17	0.00	0.00	39	0.08	0.00	0.00
12	0.00	0.00	0.00	40	0.17	0.00	0.00
13	0.08	0.00	0.00	41	2.17	0.00	0.00
14	0.17	0.00	0.00	42	1.42	0.00	0.00
15	0.08	0.00	0.00	43	0.08	0.00	0.00
16	0.08	0.00	0.00	44	0.17	0.00	0.00
17	0.08	0.00	0.00	45	1.08	0.00	0.00
18	0.25	0.00	0.00	46	0.58	0.00	0.00
19	0.33	0.56	0.00	47	0.08	0.00	0.00
20	0.50	0.00	0.00	48	0.33	0.00	0.00
21	0.75	0.00	0.00	49	0.08	0.00	0.00
22	1.08	0.22	0.25	50	0.00	0.00	0.00
23	1.08	0.00	0.00	51	0.33	0.00	0.00
24	0.75	0.00	0.00	52	0.17	0.00	0.00
25	0.75	0.11	0.00				
26	0.33	0.11	0.00	Average			
27	0.58	0.00	0.00	1969	0.44	0.03	0.01
28	0.08	0.00	0.00	1968	0.79	0.57	0.38

[a]Detection limit - 10^4 d/24 hr per particle.

Table 5.4 Radioparticulate Fallout—1969
(Gummed Paper Data—Weekly Average)

Station Number	Location	Long-Lived Activity 10^{-4} μCi/ft²	No. of Particles by Activity Ranges[a]				Total Particles Per Sq. Ft.
			10^4–10^5 d/24 hr	10^5–10^6 d/24 hr	10^6–10^7 d/24 hr	> 10^7 d/24 hr	
			Laboratory Area				
HP-1	S 3587	0.48	0.23	0.00	0.00	0.00	0.23
HP-2	NE 3025	0.94	0.79	0.06	0.00	0.00	0.85
HP-3	SW 1000	0.47	0.08	0.00	0.00	0.00	0.08
HP-4	W Settling Basin	0.65	0.21	0.00	0.00	0.00	0.21
HP-5	E 2506	0.72	0.60	0.00	0.00	0.00	0.60
HP-6	SW 3027	1.65	1.17	0.02	0.00	0.02	1.21
HP-7	W 7001	0.42	0.10	0.00	0.00	0.00	0.10
HP-8	Rock Quarry	0.37	0.06	0.00	0.00	0.00	0.06
HP-9	N Bethel Valley Rd.	0.38	0.10	0.00	0.00	0.00	0.10
HP-10	W 2075	2.17	1.47	0.02	0.00	0.02	1.51
HP-16	E 4500	0.50	0.29	0.00	0.00	0.02	0.31
HP-20	HFIR	0.32	0.02	0.00	0.00	0.00	0.02
Average		0.75	0.42	0.01	0.00	0.01	0.44
			Perimeter Area				
HP-31	Kerr Hollow Gate	0.43	0.00	0.00	0.00	0.00	0.00
HP-32	Midway Gate	0.51	0.12	0.00	0.00	0.00	0.12
HP-33	Gallaher Gate	0.37	0.04	0.00	0.00	0.00	0.04
HP-34	White Oak Dam	0.36	0.02	0.00	0.00	0.00	0.02
HP-35	Blair Gate	0.37	0.00	0.00	0.00	0.00	0.00
HP-36	Turnpike Gate	0.41	0.04	0.00	0.00	0.00	0.04
HP-37	Hickory Creek Bend	0.34	0.00	0.00	0.00	0.00	0.00
HP-38	E EGCR	0.33	0.00	0.00	0.00	0.00	0.00
HP-39	Townsite	0.45	0.02	0.00	0.00	0.00	0.02
Average		0.40	0.03	0.00	0.00	0.00	0.03
			Remote Area				
HP-51	Norris Dam	0.34	0.04	0.00	0.00	0.00	0.04
HP-52	Loudoun Dam	0.29	0.00	0.00	0.00	0.00	0.00
HP-53	Douglas Dam	0.28	0.00	0.00	0.00	0.00	0.00
HP-54	Cherokee Dam	0.30	0.00	0.00	0.00	0.00	0.00
HP-55	Watts Bar Dam	0.29	0.00	0.00	0.00	0.00	0.00
HP-56	Great Falls Dam	0.31	0.02	0.00	0.00	0.00	0.02
HP-57	Dale Hollow Dam	0.34	0.00	0.00	0.00	0.00	0.00
HP-58	Knoxville	0.36	0.00	0.00	0.00	0.00	0.00
Average		0.31	0.01	0.00	0.00	0.00	0.01

[a]Detection limit – 10^4 d/24 hr per particle.

Table 5.5

Discharge of ^{131}I from ORNL Stacks
And Weekly Average Concentration of ^{131}I in Air and Milk—1969

Week No.	PAM's Air Conc. Units of 10^{-12} μCi/cc	Milk Conc. pCi/l	Stack Discharge Millicuries[a]	Week No.	PAM's Air Conc. Units of 10^{-12} μCi/cc	Milk Conc. pCi/l	Stack Discharge Millicuries[a]
1	0.007	5	10	29	0.017	5	126
2	0.011	5	28	30	0.017	5	44
3	0.009	5	41	31	0.011	10	120
4	0.007	5	91	32	0.016	5	97
5	0.020	5	555	33	0.006	5	34
6	0.101	5	4799	34	0.006	5	23
7	0.014	5	244	35	0.009	5	98
8	0.014	5.9	226	36	0.013	5	218
9	0.014	5	104	37	0.012	5	176
10	0.010	5	214	38	0.008	5	69
11	0.010	5	136	39	0.023	6.6	182
12	0.011	5	112	40	0.017	7.8	66
13	0.016	5	230	41	0.013	10.3	57
14	0.017	5	252	42	0.009	7.4	181
15	0.021	5.9	220	43	0.015	6.6	107
16	0.018	5	384	44	0.016	6.4	190
17	0.051	8.4	1034	45	0.015	9.9	148
18	0.019	5	410	46	0.012	7.2	56
19	0.033	5	310	47	0.013	*	35
20	0.024	11.1	424	48	0.030	5	838
21	0.032	6.2	481	49	0.009	5	136
22	0.022	10	270	50	0.016	7.1	209
23	0.035	5	310	51	0.039	*	615
24	0.016	5	599	52	0.010	5	284
25	0.011	5	467				
26	0.015	7	193	Total			16,385
27	0.012	5	67				
28	0.011	5	65	Average	0.018	5.9	

[a]Data furnished by Laboratory Facilities Department.

*No samples collected.

Table 5.6 Concentration of ^{131}I in Air—1969
(Weekly Average)

Location	Units of 10^{-12} μCi/cc		
	Maximum	Minimum[a]	Average
ORNL Plant Area	5.28	0.010	0.311
Perimeter Area	0.49	< 0.010	0.018

[a]Minimum detectable amount of ^{131}I is 20 d/m. At the average sampling rate this corresponds to approximately 0.010×10^{-12} μCi/cc on the perimeter monitors and approximately 0.020×10^{-12} μCi/cc on the Plant monitors. In averaging, one-half of this value, 10 d/m, is used for all samples showing a total amount of ^{131}I less than 20 d/m.

Table 5.7 Concentration of Radioactive Materials in Rainwater—1969
(Weekly Average by Stations)

Station Number	Location	Activity in Collected Rainwater, μCi/ml
	Laboratory Area	
HP-7	West 7001	0.35×10^{-7} μCi/ml
	Perimeter Area	
HP-31	Kerr Hollow Gate	0.33×10^{-7} μCi/ml
HP-32	Midway Gate	0.31
HP-33	Gallaher Gate	0.34
HP-34	White Oak Dam	0.38
HP-35	Blair Gate	0.39
HP-36	Turnpike Gate	0.36
HP-37	Hickory Creek Bend	0.28
HP-38	E EGCR	0.43
HP-39	Townsite	0.27
Average		0.34×10^{-7} μCi/ml
	Remote Area	
HP-51	Norris Dam	0.60×10^{-7} μCi/ml
HP-52	Loudoun Dam	0.46
HP-53	Douglas Dam	0.57
HP-54	Cherokee Dam	0.56
HP-55	Watts Bar Dam	0.38
HP-56	Great Falls Dam	0.41
HP-57	Dale Hollow Dam	0.46
HP-58	Knoxville	0.33
Average		0.47×10^{-7} μCi/ml

Table 5.8

Weekly Average Concentration of Radioactivity in Rainwater—1969

Units of 10^{-7} μCi/ml

Week Number	LAM's	PAM's	RAM's	Week Number	LAM's	PAM's	RAM's
1	0.38	0.37	0.44	29	No Rain	No Rain	0.27
2	0.51	0.41	0.92	30	0.12	0.37	0.69
3	0.04	0.02	0.08	31	1.50	1.26	1.48
4	0.06	0.18	0.35	32	0.34	0.89	0.87
5	0.15	0.11	0.09	33	0.20	0.21	0.47
6	0.19	0.21	0.24	34	0.45	0.30	0.81
7	0.17	0.13	0.20	35	No Rain	No Rain	0.22
8	0.16	0.29	0.42	36	0.20	0.24	0.19
9	0.26	0.19	0.88	37	No Rain	No Rain	No Rain
10	0.27	0.44	0.53	38	0.03	0.10	0.19
11	No Rain	0.07	No Rain	39	0.05	0.14	0.15
12	0.25	0.27	0.39	40	0.12	0.21	0.48
13	0.63	0.49	0.63	41	0.05	0.12	0.18
14	0.48	0.50	0.60	42	No Rain	No Rain	No Rain
15	0.59	0.47	0.55	43	0.08	0.11	0.14
16	0.40	0.36	0.42	44	0.04	0.07	0.09
17	No Rain	No Rain	No Rain	45	No Rain	No Rain	No Rain
18	0.28	0.52	0.48	46	0.31	0.24	0.85
19	1.23	1.20	1.08	47	0.35	0.43	0.48
20	0.76	0.53	0.37	48	No Rain	No Rain	No Rain
21	1.29	1.15	1.07	49	0.06	0.09	0.12
22	0.72	1.04	1.40	50	0.09	0.09	0.27
23	0.74	1.28	1.67	51	0.05	0.10	0.24
24	0.79	0.86	0.75	52	0.12	0.13	0.15
25	0.32	0.50	0.80				
26	0.54	0.37	0.49	Average			
27	0.48	0.43	0.71	1969	0.35	0.34	0.47
28	1.15	0.66	0.63	1968	0.27	0.34	0.33

Table 5.9 Liquid Waste Discharged from White Oak Creek—1969

	Curies	
	Total for Year	Monthly Average
Beta Activity other than Tritium	13	1.08
Tritium	12,247	1021
Transuranic Alpha Emitters	0.2	0.017

Table 5.10 Yearly Discharges of Radionuclides to Clinch River (Curies)

Year	^{137}Cs	^{106}Ru	^{90}Sr	TRE*(-Ce)	^{144}Ce	^{95}Zr	^{95}Nb	^{131}I	^{60}Co	^{3}H
1965	2.1	69	3.4	5.9	0.1	0.33	0.33	0.20	12	
1966	1.6	29	3.0	4.9	0.1	0.67	0.67	0.24	7	3093
1967	2.7	17	5.1	8.5	0.2	0.49	0.49	0.91	3	13273
1968	1.1	5	2.8	4.4	0.03	0.27	0.27	0.31	1	9685
1969	1.4	1.7	3.1	4.6	0.02	0.18	0.18	0.54	1	12247

*Tri-Valent Rare Earths.

Table 5.11 Radioactivity in Clinch River—1969

Location	Concentration of Radionuclides of Primary Concern Units of 10^{-8} μCi/ml							% $(MPC)_w$
	^{90}Sr	^{144}Ce	^{137}Cs	$^{103-106}Ru$	^{60}Co	$^{95}Zr-^{95}Nb$	^{3}H	
CRM 23.1[a]	0.08	0.03	0.09	0.08	0.38	0.04	*	0.30
CRM 20.8[b]	0.09	< 0.01	0.04	0.04	0.03	< 0.01		0.36
							323	0.11
CRM 4.5[a]	0.11	0.02	0.19	0.14	0.35	0.04	*	0.41

[a]Measured values.

[b]Values given for this location are calculated values based on the concentrations of wastes released from White Oak Dam and the dilution afforded by the Clinch River; they do not include radioactive materials (e.g., fallout) that may enter the river upstream from CRM 20.8.

*No analysis.

Table 5.12

Calculated Percent $(MPC)_w$ of Radioactivity in Clinch River Water Below the Mouth of White Oak Creek—1969

Month	% MPC_w Tritium	% MPC_w Other Radionuclides
January	0.21	0.34
February	0.24	0.58
March	0.27	0.50
April	0.35	0.79
May	0.12	0.39
June	0.09	0.33
July	0.02	0.23
August	0.05	0.35
September	0.03	0.35
October	0.08	0.47
November	0.08	0.21
December	0.12	0.41
Weighted Average		
1969	0.11	0.36
1968	0.07	0.83

Table 5.13

Weekly Average Concentration of ^{90}Sr in Raw Milk
In the Immediate Environs of Oak Ridge—1969

Week Number	pCi/l	Week Number	pCi/l
1	18.4	29	19.8
2	20.0	30	19.0
3	22.5	31	19.2
4	17.7	32	17.7
5	23.4	33	17.4
6	19.3	34	19.3
7	18.5	35	14.4
8	24.6	36	18.6
9	24.9	37	16.3
10	19.7	38	18.2
11	21.6	39	15.2
12	20.6	40	18.3
13	20.9	41	9.4
14	20.8	42	9.3
15	18.1	43	*
16	18.4	44	8.8
17	19.8	45	9.1
18	21.1	46	9.0
19	19.5	47	*
20	22.7	48	7.7
21	23.8	49	11.2
22	18.6	50	9.2
23	20.9	51	*
24	17.2	52	9.8
25	17.5		
26	27.9		
27	18.2		
28	19.7	Average	17.8

*No samples collected.

Table 5.14

Background Radiation in East Tennessee Area—1969

Stations	μR/hr
Great Falls	11.1
Dale Hollow	11.8
Crossville	11.3
Watts Bar	14.4
Rockwood	11.4
Wartburg	10.2
Kingston	12.5
Oliver Springs	11.8
Lenoir City	10.4
Clinton	11.7
Norris	11.6
Powell	14.7
Halls Cross Roads	13.7
Strawberry Plains	12.4
Cherokee	11.3
AVERAGE	12.0

Table 5.15

Average Gamma Count Rate at the Surface of Clinch River Silt

Location of Cross Section	Gamma Activity - Counts Per Second (48 c/s = ~ 0.03 mR/hr		
	1966	1967	1969
CRM 51.6	*	*	8.1
50.5	Entry of Braden Branch		
48.8	*	*	19.2
42.8	7.3	7.8	14.8
39.1	*	*	18.4
34.7	7.4	7.0	20.1
31.1	6.8	6.8	18.2
29.0	*	*	18.7
27.0	*	*	15.2
24.9	6.7	7.3	20.5
23.1	Melton Hill Dam		
21.5	5.0	2.9	2.4
20.8	Entry of White Oak Creek		
19.1	*	8.0	7.4
17.8	*	11.9	11.9
16.3	35.1	16.8	12.0
15.2	*	28.3	12.7
14.0	37.3	28.9	28.7
12.5	*	13.5	11.5
11.0	63.1	48.0	30.9
9.4	*	19.7	18.4
8.0	68.8	60.1	51.5
5.8	67.8	57.0	56.0
4.7	46.9	36.7	37.3
1.1	46.6	42.0	39.8

*No measurements taken.

NOTE: No survey made in 1968.

Table 5.16

Average Gamma Count Rate at the Surface of Tennessee River Silt

Location of Cross Section	Gamma Activity - Counts Per Second (48 c/s = ~ 0.03 mR/hr		
	1966	1967	1969
TRM 570.8	8.9	8.7	8.3
567.6	Entry of Clinch River		
562.7	17.3	16.3	15.1
552.7	17.9	12.6	16.4
543.8	12.8	15.5	11.6
532.0	11.5	11.5	11.2
530.0	Watts Bar Dam		
509.5	*	*	6.5
491.9	12.6	*	11.5
475.1	*	*	10.7
471.0	Chickamauga Dam		
434.1	*	*	7.8
425.2	*	*	8.6
425.0	Nickajack Dam		
404.0	*	*	3.2
381.2	*	*	8.9
354.4	9.5	*	8.8
349.0	Guntersville Dam		

*No measurements taken.

NOTE: No survey made in 1968.

6.0 PERSONNEL MONITORING

It is the policy of the Oak Ridge National Laboratory to monitor the radiation exposure of all persons who enter Laboratory areas where there is a likelihood of radiation exposure. Dose analysis is accomplished mainly through the use of personnel meters, bio-assays, and in vivo counting (whole body counter) techniques.

6.1 Dose Analysis Summary, 1969

6.1.1 External Exposures - No employee received a whole body radiation dose which exceeded the maximum permissible levels recommended by the Federal Radiation Council (FRC). The highest whole body dose received by an employee was about 3.8 rem or 32 percent of the maximum permissible annual dose. The range of doses for persons using ORNL badge-meters is shown in Table 6.1.

As of December 31, 1969, no employee had a cumulative whole body dose which exceeded the recommended maximum permissible dose as based on the age proration formula 5(N - 18) (Table 6.2). Only one employee had an average annual exposure rate that exceeded 5 rem per year of employment (Table 6.3).

The highest cumulative dose to the skin of the whole body received by an employee during 1969 was about 8.8 rem or 29 percent of the maximum permissible annual skin dose of 30 rem.

As of December 31, 1969, the highest cumulative dose of whole body radiation received by an employee was approximately 93 rem. This dose was accrued over an employment period of about 26 years and represented an average exposure of about 3.6 rem per year.

The highest cumulative hand exposure recorded during 1969 was about 33 rem or 44 percent of the recommended maximum permissible annual dose to the extremities.

The average of the ten highest whole body doses of ORNL employees for each of the years 1965 through 1969 are shown in Table 6.4. The highest individual dose for each of those years is shown also.

The average annual dose to ORNL employees for the years 1965 through 1969 is the subject of Table 6.5. This rather arbitrary quantity is obtained by dividing the sum of all doses for the year by the number of employees involved.

6.1.2 Internal Exposure - During 1969 there were no cases of internal exposure where the deposition of radioactive materials within the body was estimated to have averaged greater than one-half a maximum permissible body burden.[4]

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[4] NBS Handbook 69 values are the basis for these determinations.

Three employees continued to have estimated body burdens of transuranic alpha emitters (mainly ^{239}Pu) of 35 to 45 percent of the recommended maximum permissible value.[5] The ICRP recommends, Publication 6, paragraph 86(a), individuals who exceed 50 percent of a maximum permissible body burden be placed on a work assignment where the potential for internal exposure is reduced.

6.2 External Dose Techniques

6.2.1 Film Meters - Film meters are issued to all persons who have access to ORNL facilities in which there is a likelihood of radiation exposures for which monitoring is required. Photo-badge-meters are assigned to all ORNL employees, and to certain other persons who are authorized to enter ORNL facilities. Temporary meters may be issued in lieu of photo-badge-meters for short-term use.

NTA (nuclear track) film packets are included in all film meters. The NTA films are processed routinely if the badge-meter is assigned to an individual who normally works where there may be exposure to neutrons; otherwise the films would be processed only in the event of a nuclear accident.

Beta-gamma sensitive films from badge-meters issued to full-time employees are processed routinely each calendar quarter (or more frequently if necessary). Films used in other meters are processed as conditions of use may require. Films from meters issued to visitors are processed if there is a likelihood that a radiation exposure was incurred.

High-level radiation dosimetry components of the badge meters (sulfur, gold, indium, and metaphosphate glass) are for use in the event that doses exceed the capability of the monitoring films.

6.2.2 Pocket Meters - Pocket meters (indirect reading, ionization chambers) are made available at all principal points of entry to ORNL premises. A pair of pocket meters are carried for the duration of a work shift by persons who work in an area where the potential for an exposure of 20 mR or more exists during the work shift. Pocket meter pairs are processed each day by Health Physics technicians and readings of 20 mR or more are reported daily to supervision. Pocket meters are used for a day-to-day record of integrated exposure.

6.2.3 Hand Exposure Meters - Hand exposure meters are film-loaded finger rings used to measure hand exposure. Hand exposure meters are issued to persons for use during operations where it is likely that the hand dose is such as to exceed 1 rem

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[5] AEC Manual Chapter 0502 requires an evaluation of the radiation exposure status of an employee when monitoring techniques indicate that a body burden equals or exceeds 50 percent of a maximum permissible limit.

during the week. They are issued and collected by Radiation and Safety Surveys Section personnel who determine the need for this type of monitoring and arrange for a processing schedule.

6.2.4 Metering Resumé - Shown in Table 6.6 are the quantities of personnel metering devices used and processed during 1969. The number of films processed is less than the number issued, because those which are issued for accident dosimetry only are not processed unless there is a likelihood of exposure.

6.3 Internal Dose Techniques

6.3.1 Bio-Assays - Urine and fecal samples are analyzed for the purpose of making internal dose determinations. The frequency of sampling and the type of radiochemical analysis performed are based upon each specific radioisotope and the exposure potential. Because of the small quantities of radioactive material in most samples, qualitative analyses are not feasible, and only quanitative analyses for predetermined isotopes are performed routinely.

In most cases bio-assay data require interpretation to determine the dose to the person; computer programs are used for evaluation of extensive data on urinary excretion of ^{239}Pu. An estimate of dose is made for all cases in which it appears that one-third of a body burden, averaged over a calendar year, may be exceeded.

6.3.2 Whole Body Counter - The whole body counter (an in vivo gamma spectrometer) may be used for determining internally deposited quantities of most of the gamma ray-emitting substances and some of the more energetic beta-emitting substances. Thus, it provides a direct method of determining body burdens of those substances.

6.4 Records and Reports

Most records and reports are prepared by automatic data processing (ADP) techniques through the use of high-speed digital computer systems. The IBM 7090, located at the Computing Technology Center (CTC), provides routine weekly, quarterly, and annual reports involving external dose data. A typical weekly report is shown in Table 6.7; a typical quarterly report is shown in Table 6.8. An IBM 360, operated by the ORNL Math Panel, is used to prepare the weekly pocket meter report (Table 6.9) as well as the weekly, quarterly, and annual bio-assay reports. A sample of the Weekly Bio-Assay Sample Status Report is shown in Table 6.10.

A quarterly and an annual report (Table 6.11) based on results of analysis by the whole body counter (IVGS) are prepared by the IBM 360 at CTC.

A quarterly and an annual report of occupational injuries are processed by the IBM 360 at ORNL.

An individual external dose summary (Table 6.12) is prepared annually be updating on the IBM 360 at CTC.

Body burden estimates of ^{239}Pu are prepared in report form (usually quarterly) by use of the IBM 7090 at CTC.

Permanent files are maintained at Applied Health Physics and Safety Headquarters for each individual who is assigned an ORNL photo-badge-meter. An IBM card cross-indexing system is maintained at the principal monitoring stations for the purpose of expediting meter assignments. These IBM cards are compatible with the various computer programs and provide for the internal audit of all personnel monitoring record data.

Copies of the ADP reports, both temporary and final, are maintained for both the internal and external dose programs. Data used in the ADP program are stored on computer-quality magnetic tapes. Data pertinent to the work of the dosimetry groups and information used in the non-ADP reports are maintained in record form by the Dose Data Group.

Table 6.1 Dose Data Summary for Laboratory Population Involving Exposure to Whole Body Radiation—1969

Group	Number of Rem Doses in Each Range							Total
	0-1	1-2	2-3	3-4	4-5	5-6	6 up	
ORNL Employees	5443	90	25	2	0	0	0	5560
ORNL-Badged Non-Employees	236	0	0	0	0	0	0	236
TOTAL	5679	90	25	2	0	0	0	5796

Table 6.2 Average Rem Per Year Since Age 18—1969

	Number of Doses in Each Range				Total
	0-2.5	2.5-5.0	5.0-7.5	7.5 up	
ORNL Employees	5554	6	0	0	5560

Table 6.3 Average Rem Per Year of Employment at ORNL—1969

	Number of Doses in Each Range				Total
	0-2.5	2.5-5.0	5.0-7.5	7.5 up	
ORNL Employees	5536	23	1	0	5560

Table 6.4 Average of the Ten Highest Whole Body Doses and the Highest Individual Dose by Year

Year	Average of the Ten Highest Doses (Rem)	The Highest Dose (Rem)
1965	3.50	4.41
1966	3.81	4.85
1967	4.01	5.10
1968	4.11	4.71
1969	2.84	3.79

Table 6.5 Average Annual Whole Body Dose to the Average ORNL Employee

Year	Average Dose (Rem)
1965	0.123
1966	0.126
1967	0.142
1968	0.114
1969	0.088

Table 6.6 Personnel Meter Services

	1967	1968	1969
A. Pocket Meter Usage			
1. Number of Pairs Used			
ORNL	150,748	143,572	128,024
CPFF	19,344	5,564	7,228
Total	170,092	149,136	135,252
2. Average Number of Users per Quarter			
ORNL	1,408	1,273	1,149
CPFF	252	94	120
Total	1,660	1,367	1,269
B. Film Usage			
1. Films Used in Photo-Badge-Meters			
Beta-Gamma	20,800	22,100	20,930
NTA	10,300	10,940	10,360
2. Films Used in Temporary Meters			
Beta-Gamma	4,930	8,850	8,440
NTA	1,600	2,860	2,730
C. Films Processed for Monitoring Data			
1. Beta-Gamma	21,150	22,720	21,800
2. NTA	1,580	1,190	1,400
3. Hand Meter	2,490	1,110	1,100

Table 6.7 Typical ORNL Film Monitoring Data

Name	ID Number	Symbol	Dosimetry Dates Wk-Yr	Dosimetry Dates Qtr-Yr	Meter Dose DS	Meter Dose DC
Last Name, Initials	PR. No.	PF	35-63	3-63	0.000	0.000
Last Name, Initials	PR. No.	PF	31-63	3-63	0.120	0.090
Last Name, Initials	PR. No.	PF	30-63	3-63	0.030	0.000
Last Name, Initials	PR. No.	PF	36-63	3-63	0.070	0.020
Last Name, Initials	PR. No.	PF	34-63	3-63	0.000	0.000
Last Name, Initials	PR. No.	PF	36-63	3-36	0.370	0.310
Last Name, Initials	PR. No.	PF	32-63	3-63	0.000	0.000
Last Name, Initials	PR. No.	PF	33-63	3-63	0.040	0.020
Last Name, Initials	PR. No.	PF	34-63	3-63	0.260	0.130
Last Name, Initials	PR. No.	PF	35-63	3-63	0.040	0.010

Table 6.8 Typical ORNL Personnel Radiation Exposure Record

Name	ID Number	Symbol	Date Wk-Yr	REM DS	REM DC	REM This Qtr DS	REM This Qtr DC	REM This Yr DS	REM This Yr DC	Total REM DC	A	DC/A
----	-----	PF	39-63	0.860	0.630	0.860	0.630	1.68	1.32	35.59	18	2.02
		PN	39-63	0.000	0.000							
----	-----	PF	39-63	0.340	0.240	0.340	0.240	0.34	0.24	0.24	1	0.80
----	-----	PF	39-63	0.020	0.010	0.020	0.010	0.02	0.01	5.21	14	0.38
----	-----	PF	39-63	0.070	0.040	0.070	0.040	0.30	0.19	18.38	16	1.19
----	-----	PF	39-63	0.390	0.310	0.390	0.310	1.40	1.14	2.74	20	0.14
----	-----	PF	39-63	0.350	0.150	0.350	0.150	0.77	0.49	9.60	17	0.56
----	-----	PEL	27-63	0.010	0.010	0.150	0.120	0.27	0.24	5.55	6	1.09
		PF	39-63	0.140	0.110							
		PN	39-63	0.000	0.000							
----	-----	PF	39-63	0.400	0.200	0.400	0.200	0.73	0.45	7.43	12	0.64
----	-----	PF	39-63	0.180	0.150	0.180	0.150	0.60	0.49	8.43	7	1.34
----	-----	PF	39-63	0.330	0.110	0.360	0.140	0.81	0.34	3.00	13	0.24
		PN	39-63	0.030	0.030							
----	-----	PF	39-63	0.180	0.080	0.180	0.080	0.51	0.33	29.82	18	1.68
		PN	39-63	0.000	0.000							
----	-----	PF	39-63	0.320	0.270	0.320	0.270	1.14	0.98	22.76	13	1.76
		PN	39-63	0.000	0.000							
----	-----	PF	39-63	0.420	0.290	0.420	0.290	1.85	1.11	15.86	16	1.04
----	-----	PF	39-63	0.320	0.140	0.320	0.140	0.67	0.46	8.96	11	0.84
----	-----	PF	39-63	0.390	0.210	0.390	0.210	1.21	0.72	33.62	18	1.87

Table 6.9 Typical Pocket Meter Weekly Report

DEPT XXXX	HP WK 52														
NAME	PR NO	DC	S	M	T	W	T	F	S	WK	QTR	F	SMB	DCP/PMD	BAL
---------------	-----			0	5	10	0			15	270	62			
---------------	-----	660		10	25		120			155	1140	57	DWQ	660/700	440
---------------	-----										35	11			
---------------	-----										10	1			
---------------	-----			10	10	0	10			30	220	41			
---------------	-----			10						10	115	62			
---------------	-----										5	20			
---------------	-----		0	10	10		20			40	560	54	D Q		
---------------	-----										195	22			
---------------	-----											1			
---------------	-----		125	0		0				125	260	60	DW		
---------------	-----			30	0	5	0			35	415	61	D		
					ENTRIES			+		D	W	Q	COUNT		
						23				5	2	2	12		

Table 6.10 Typical Weekly Bio-Assay Sample Status Report

RESULTS THIS REPORT 12-20-65

Div. Code	Name	PR NO	HP AREA Number	Type Analysis	Receipt Date	Type Sample	Sample Priority	d/m/Sample	d/m/24 hrs
HP	----------	-----	3550	GAO	12-16-65	U	3		0
HP	----------	-----	3019	PUO	12-12-65	U	3		0
HP	----------	-----	3019	PUO	12-16-65	U	3		0
HP	----------	-----	3019	SRO	12-12-65	U	3		0
Div. Total 4									

Table 6.11 Typical Whole Body Counter (IVGS) Summary Report

HEALTH PHYSICS WHOLE BODY COUNTER IVGS
ANALYSES FOR
1968

NAME	ID NO	-------------DIVISION DIV	HP-AREA	DATE	REASON	TYP-SER-NO	RESULTS	REMARKS
------------	------	---	----	07-10-68	WORK AREA	SC-24732	NORMAL HUMAN SPECTRUM	
------------	------	---	----	03-05-68	INCIDENT	CH-24115	NORMAL HUMAN SPECTRUM	
------------	------	---	----	03-05-68	INCIDENT	SC-24116	NORMAL HUMAN SPECTRUM	
------------	------	---	----	05-28-68	WORK AREA	SC-24500	LESS THAN 10PCT MPBB	
------------	------	---	----	01-10-68	FOLLOW-UP	SC-23859	LESS THAN 25PCT MPBB	
------------	------	---	----	02-07-68	FOLLOW-UP	CH-23996	LESS THAN 50PCT MPBB	
------------	------	---	----	02-07-68	FOLLOW-UP	SC-23995	LESS THAN 25PCT MPBB	
------------	------	---	----	03-05-68	INCIDENT	CH-24113	LESS THAN 50PCT MPBB	90SR 30PCT LUNG BURDEN
------------	------	---	----	03-05-68	INCIDENT	SC-24112	LESS THAN 10PCT MPBB	
------------	------	---	----	06-25-68	FOLLOW-UP	SC-24652	LESS THAN 25PCT MPBB	
------------	------	---	----	05-28-68	WORK AREA	SC-24506	NORMAL HUMAN SPECTRUM	

Table 6.12 Typical Individual External Dose Summary

Name - Employee AN

		Symbol	Definition
I.D. Number	5782	DC	Cumulative recorded total rem to whole body since activation date.
S.S. Number	221-16-0038		
Birth Date	6/17/28	DO	Dose data other than that reported herein Yes (3)
Activation Date	1/16/48		

Year	QTR	Rem for Qtr Skin	Rem for Qtr Body	Rem for Year Skin	Rem for Year Body	Rem DC
				DC Prior to 1961		22.13
1961	1	.26	.19			
	2	.20	.16			
	3	.29	.12			
	4	.44	.36			
Total				1.30	.83	22.96
1962	1	.33	.30			
	2*	.56	.48			
	3*	.69	.54			
	4	.59	.51			
Total				2.17	1.83	24.79
1963	1	.61	.50			
	2	.53	.43			
	3	.78	.43			
	4	.03	.03			
Total				1.95	1.39	26.18
1964	1	.04	.03			
	2	.02	.01			
	3	.02	.01			
	4	.09	.04			
Total				.17	.09	26.27
1965	1	.25	.12			
	2	.40	.22			
	3	.48	.28			
	4	.41	.21			
Total				1.54	.83	27.10

7.0 LABORATORY OPERATIONS MONITORING

Radiation incidents are classified according to a severity index system developed over the past several years.[6] The method serves to index unusual occurrences according to degree of severity and permits a system of analysis regarding Applied Health Physics and Safety practices among Laboratory operations. This report summarizes the unusual occurrence frequency rate and discusses some of the problems encountered among Laboratory facilities.

7.1 Unusual Occurrences

During 1969 there were 12 unusual occurrences recorded which represents a decrease of 40 percent over the number reported for 1968 (Table 7.1). The number for 1969, twelve, is approximately 45 percent below the five-year average of 22 for the years 1965 through 1969. The frequency rate of unusual occurrences among Laboratory divisions involved (Table 7.2) is known to vary in relationship to the quantity of radioactive materials handled, the number of radiation workers involved, and the radiation hazard potential associated with a particular operation or facility.

Seven of the incidents reported during 1969 involved area contamination that was handled by the regular work staff without appreciable production or program loss. One incident required the partial shut-down of a facility and several man-hours of effort were expended during the cleanup campaign. Five occurrences involved personnel contamination requiring decontamination under medical supervision, and one incident involved the hand exposure to an employee that required imposition of minor work restrictions.

7.2 Radiation Surveys

During 1969 Radiation and Safety Surveys personnel assisted the operating groups in keeping the contamination, air concentration, and personnel exposure levels well below the established maximum permissible limits. Through seminars, safety meetings and informal discussions with supervision, they assisted in reducing or eliminating a number of problems associated with radiation protection at the Laboratory. The following is a brief description of some of the problems and methods of solution.

7.2.1 Transfer of Uranium to the TRUST Facility, Building 3019 - Health Physics assisted the Chemical Technology Division in developing procedures for and provided on-the-job surveillance of the transfer of 1047 Kg uranium (75% ^{235}U, 11% ^{233}U) to the TRUST (Thorium Reactor Uranium Storage Tank) Facility. The TRUST Facility, completed in 1968, was designed to receive, transfer and store indefinitely the highly

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[6] See ORNL-3665, Applied Health Physics Annual Report for 1968, pp. 14-15.

enriched uranium from the Indian Point reactor fuel. Nuclear Fuel Services, West Valley, New York, processed the fuel. Purified uranyl nitrate solution was shipped to ORNL in drums containing 10-liter "safe" plastic bottles. A total of 24 shipments commencing December 16, 1968, and ending February 6, 1969, was required to transfer a total of 527 containers.

Exposure rates near the surface of the plastic bottles ranged from 200 mR/hr to 600 mR/hr due to the ^{232}U content (120 parts ^{232}U per 10^6 parts uranium). With careful monitoring and the cooperation of all personnel involved the transfers were completed with no significant release of contamination; with an average exposure for the period of $\sim$20 mR/man-day; and the highest exposure for the period was 650 mR.

Air sparging of the storage tank prior to subsequent monthly samplings resulted in high radiation levels in the TRUST off-gas system due primarily to deposition of ^{220}Rn daughters at a pressure control valve and on the off-gas filters. Temporary Radiation Zones were established to control access to these areas until the activity decayed to safe levels. An entrainment separator and an SGN mini-caisson high efficiency filter have been installed to reduce this problem.

7.2.2 Renovation of Cell 7 of the High Radiation Laboratory Analytical Facility (HRLAF), Building 3019 - Health Physics consultation and on-the-job surveillance were provided during renovation of HRLAF Cell 7 for installation of a spectrophotometer facility. Removal of the old cell pan and equipment, decontamination, and application of shielding on residual "hot" spots reduced cell radiation levels from > 10 rad/hr to < .5 rad/hr. Numerous entries into the cell were through a wood-framed plastic tent which greatly enhanced contamination control and minimized the effects on the ventilation of adjacent cells. The installation was completed with no spread of contamination outside established control zones and with exposures to personnel well within permissible limits.

7.2.2(a) Additional Hurst Threshold Detector Units Installed, Building 3019 - The quantities of fissile material in storage or in process in the Radiochemical Pilot Plant, Building 3019, have increased considerably. The Chemical Technology Division, implementing recommendations by Health Physics, installed additional Hurst Threshold Detector Units in the Penthouse, Cell 4, Room 211, Room 502, and Building 3100. All units were installed in recovery tubes accessible from outside the buildings.

7.2.3 Applied Health Physics and Safety Assistance During a Series of Hot Cell Experiments in Building 4507 - Applied Health Physics and Safety personnel provided radiation monitoring assistance during a series of six hot cell experiments made in Cell 2 of Building 4507 in which short-decay stainless steel-clad, sol-gel 15 percent PuO_2—85 percent UO_2, which had been irradiated to about 20,000 Mwd/ton (U + Pu), was processed. A gloved box with bag-out facilities was installed at the cell opening for the removal of gas and liquid samples. All cell openings were monitored for the removal of samples and waste materials. The experiments were completed without significant contamination spread or personnel exposure.

7.2.4 Addition of Shielding Studies Facility at Building 6010 – A shielding studies facility was added to ORELA during calendar year 1969. The facility utilizes the 165° flight path and is located on the northwest corner of Building 6010. Preliminary radiation measurements have been made but more meaningful results will be obtained when the final geometry of the beam and arrangement of the experimental equipment has been determined.

7.2.5 ORELA Acceptance Phase for Operational Purposes – The Oak Ridge Linear Accelerator (ORELA) was accepted for use and became operational during 1969. Detailed radiation surveys made while the machine was delivering near maximum beam current indicate the design shielding is effective in reducing the penetrating radiation generated by the machine to acceptable levels in all areas other than where beams of radiation were intended. Those areas where elevated levels of radiation were found, outside of interlocked areas, had been anticipated and exposure to personnel in such areas are to be controlled through enforcement of limited access. Response of the stack air monitor indicates production of airborne radioactive contaminants during operation of the accelerator is also within range of predicted values and is acceptable for discharge (only gases are released, as particulate matter is filtered). Personnel monitoring data available to date indicate that a major portion of the radiation exposure sustained by personnel at ORELA will be a result of maintenance efforts during accelerator down times. Dose rates of $\sim$ 250 rad/hr at 2 inches from the side of an unshielded target have been observed eight hours after accelerator shut-down. General background in the target room with the target shielded by 4 inches of lead was observed to be as high as $\sim$ 100 mrad/hr one day after shut-down.

7.2.6 Annual Survey of X-Ray Equipment – All operating X-ray units in use at ORNL were reviewed during the year. The review focused on the following items: (1) clean and adequate identification of X-ray machine and work areas in which the machines are located, (2) radiation leakage around the machine, (3) operation and integrity of interlocks and other safety devices, (4) person primarily responsible for X-ray machine and designated operators for same, (5) changes in equipment or experiments since the last review, (6) blueprints and/or diagrams of safety equipment, (7) written operating procedures. Progress toward standardization in items 1, 3, 6, and 7 continues with considerable emphasis on 6 and 7.

7.2.7 Applied Health Physics and Safety Activities in the Transuranium Research Laboratory during 1969 – The TRL staff of five Applied Health Physics and Safety personnel continued to collaborate with TRL researchers, working directly with them in planning and conducting specific experiments with transuranium isotopes at the TRL.

In addition, TRL Health Physics personnel continued their functions of inspecting, testing, and operating all stages of the building containment and air cleaning systems. In this regard the policy of containing significant quantities of radioactive material in approved enclosures and the requirement for monitoring at appropriate times at the enclosure continue to be observed. Forty enclosures are now in use including 3' and 6' gloved boxes, lead-shielded gloved boxes, concrete-shielded manipulator cells and custom-made enclosures for special apparatus.

Typical areas in which the Health Physics staff is providing experimental assistance include: preparation of organo-metallic compounds under inert atmosphere, studies involving the search for super-heavy elements in nature, chemical separation and preparation of special transuranium targets for cyclotron bombardment, research assistance at the ORIC, and microchemical CHN analyses and spectrophotometric analyses of compounds of the transuranium elements.

Other activities included the establishment of a routine program utilizing LiF-Teflon dosimeters for special monitoring problems related to TRL operations, and planning the construction of a manganese sulphate bath facility for the absolute calibration of ^{252}Cf and other neutron sources of interest to TRL researchers.

In collaboration with GE and C air handling engineers a system was designed and tested for monitoring airflows through TRL enclosures to assure appropriate ventilation and dilution in the enclosure.

Interaction with personnel from other installations regarding problems related to the handling of transuranic materials included conferences with representatives of Battelle-Northwest, Savannah River Plant, LRL, Berkeley, ORAU, N. C. State University, Rocky Flats, etc.

Studies were begun by the Health Physics staff regarding the use of Halon-1301 as a fire extinguishing agent and possible explosion suppression agent for TRL gloved boxes and cells.

7.2.8 <u>Health Physics Coverage at the Radio Isotopic Sand Tracer Project</u> - In a repeat performance of a project instituted during 1967 a representative of the Radiation and Safety Surveys Section acted as the project health physicist at the Radio Isotopic Sand Tracer Tests, conducted by the Technical Services Group of the Isotopes Division for the U. S. Corps of Engineers at Point Conception, California, and at Point Mugu, California. The tests involved placing radioactive sand (tagged with $^{198-199}$Au) off-shore in the ocean and tracing its movement along the ocean floor by use of a specially designed radiation detection system. The Health Physics representative provided on-the-job surveillance, served as custodian of radioactive material as well as assuring that all Federal and State regulations pertaining to the handling of radioactive material were followed. The tests were completed without any significant contamination or exposure problems.

7.2.9 <u>Renovation and Conversion of Cell 5, Building 3028</u> - Cell 5, a multi-kilo-curie (MKC) cell which had been used to process ^{90}Sr sources was completely stripped of all equipment. Health Physics helped develop procedures for the removal of the grossly Sr contaminated (1000 rem/hr at 3 feet) equipment and provided on-the-job surveillance during the removal of all material and subsequent decontamination of the cell. All persons working on this job were dressed in air-supplied plastic suits. Because of the high radiation readings, it was necessary to handle some of the equipment remotely. All waste was placed in lead-lined Dumpsters for transfer to the burial ground. After contamination levels were reduced to insignificant levels, new service equipment was installed and modifications were made so that an inert atmosphere could be obtained in the cell if desired.

This is the first MKC cell at ORNL to be modified in this manner. Careful planning and rigid controls played a large part in this program being completed without any significant contamination or radiation problems.

7.2.10 Removal of Shipping Barricade and Installation of Alpha Handling Facility at Building 3038 – Approximately thirty feet of the shipping barricade including the Radio Isotope storage racks were removed from Building 3038. Health Physics helped develop procedures for the removal of the grossly β,γ-contaminated (1000 rem/hr at 6 inches) concrete and provided on-the-job surveillance. An enclosure of sheet plastic was constructed around the part to be removed and all persons working inside the enclosure wore air-supplied plastic suits. A layer of wet sand was spread over the grossly contaminated floor. This served as shielding and prevented spread of contamination while the floor was chipped with jackhammers. The sand and chipped concrete were placed in metal cans, bagged in plastic, sealed and transferred to the burial ground. When the contamination levels were reduced to an insignificant level construction of the Alpha Handling Facility was begun.

This facility will contain five 5 ft. x 5 ft. stainless steel, H_2O shielded, remote handling cells. The cells are so constructed that an inert atmosphere can be obtained if desired. By careful planning this program was completed without any significant contamination or exposure problems.

7.2.11 The Decontamination and Installation of New Processing Equipment in Cell G, Building 7930 (TURF) – Cell G in Building 7930 (TURF) was decontaminated by removing the ^{233}U which had been processed in the cell for use as fuel for the MSRE. Initial entries in the cell were made with air-supplied plastic suits. The stainless steel ceiling, walls and floor were hand scrubbed and jet hosed at least three times. Airborne activity levels in the cell were brought down to well below MPC_a values enabling later entries in the cell without the need for respiratory protection.

Installation of new equipment in the cell was started in November. This equipment is to be used in the final purification of ^{252}Cf and ^{253}Cf, an extension of a TRU program.

7.2.12 Operations Involving the Molten Salt Reactor (MSRE), Building 7503 – The MSRE was successfully operated during 1969 for 4,165 equivalent full-power hours. With the exception of one relatively minor release of airborne activity in the reactor bay, surface contamination and air activity levels were kept well below permissible values throughout the building. The reactor was shut down in December and is now on standby.

7.2.13 Applied Health Physics and Safety Assistance during Experiments and Target Rod Problems at the HFIR (Building 7900) - Two major experiments were conducted in the HFIR during the year. One was a high temperature graphite fuel irradiation test and the second involved the attempted formation of ^{258}Fm in a pneumatic rabbit tube inserted into the target rod region. A relatively short-term boiling water experiment was conducted in the hydraulic rabbit position, which is the center of the fuel elements. A target rod was simulated using a tungsten rod clad in stainless steel, with boiling being detected primarily by use of thermocouples. Dose rates to personnel conducting the experiments were minimal and the experiments were completed without incident.

Cracks in the HFIR target rods recurred for the first time since the initial cracks appeared two years ago. This further contaminated the reactor primary coolant system with alpha, primarily ^{244}Cm. The target rods have been redesigned to make them more reliable, and these have withstood irradiation several times longer than any other rod.

An instrument was developed and placed in service to aid in the detection of HFIR target rod leaks. It performs its function by detecting the neutrons resulting from spontaneous fission of the transplutonium elements which leak from the target rods into the primary coolant stream. Smears and air samples taken in the building indicate that tritium and all other radioactive materials are well below allowable limits.

7.2.14 Health Physics Assistance during the Modification of Exhaust System in Building 3517; Cleanup of Cell 15 and Utilization of Cell 10 as a Storage Area in the Same Facility - The processing of multi-curie quantities of dry powder in Building 3517 continues to present unique problems in both contamination and exposure control. Modification of the exhaust system by the addition of a header through the high-bay area has improved bay containment, thus, reducing the spread of particulate activity throughout the building from a release within the bay area.

Cleanup of Cell 15 (the waste removal cell) is under way and a new waste carrier has been put into service. This carrier is a bottom loader designed to handle drums through the top of Cell 15. It is hoped that the use of this carrier will eliminate the routine loading of a stainless steel Dumpster in Cell 15 and some of the problems which were associated with this operation in the past.

Also at 3517, Cell 10 west has been utilized as a storage facility for curium and plutonium. At the close of 1969 this facility contained kg quantities of curium and $\sim$400 grams of plutonium.

7.2.15 The Renovation of Lab 4, Building 3508, in an Effort to Improve Contamination Control and Minimize Exposure to Personnel - The introduction of the transplutonium elements into Building 3508 has taxed the facility for controlling the spread of contamination and minimizing exposures. The renovation of Lab 4 was an effort to improve both. Fifteen of the original glove boxes were disconnected and removed to the burial ground with a minimum spread of contamination. The room was decontaminated to an acceptable level and the floor was given a heavy epoxy covering. The glove boxes were replaced with individual portable boxes which, while offering no better shielding, can be removed from the system once they have become so highly contaminated as to pose either a contamination or radiation hazard.

7.3 Laundry Monitoring

There were 803,330 articles of wearing apparel monitored at the laundry during 1969. Approximately eight percent were found contaminated. Of the 406,985 khaki garments monitored during the year, 207 were found contaminated. This was an increase of about 50 percent from last year.

There were 10,591 full-face respirators cleaned and monitored during the year. Of this number, 822 required additional decontamination measures prior to being placed back in service. Also, 6,814 respirator canisters were cleaned and monitored.

Table 7.1 Unusual Occurrences Summarized for the 5 -Year Period Ending with 1969

	1965	1966	1967	1968	1969
Number of Unusual Occurrences Recorded	41	22	16	20	12
A. Number of incidents of minor consequence involving personnel exposure below MPE limits and requiring little or no cleanup effort	11	8	5	7	4
B. Number of incidents involving personnel exposure above MPE limits and/or resulting in special cleanup effort as the result of contamination	30	14	11	13	8
1. Personnel Exposures	12	8	5	9	6
a. Nonreportable overexposures with minor work restrictions imposed..........	11	8	5	9	6
b. Reportable overexposures with work restrictions imposed..........	1	0	0	0	0
2. Contamination of Work Area	28	14	11	13	8
a. Contamination that could be handled by the regular work staff with no appreciable departmental program loss	27	12	11	13	7
b. Required interdepartmental assistance with minor departmental program loss..........	1	2	0	0	1
c. Resulted in halting or temporarily deterring parts of the Laboratory program	0	0	0	0	0

Table 7.2 Unusual Occurrence Frequency Rate within the Divisions for the 5-Year Period Ending with 1969

Division	No. of Unusual Occurrences 1965	1966	1967	1968	1969	5-Year Total	Percent Lab. Total (5-Year Period)
Analytical Chemistry	6	1	3	4	1	15	13.5
Biology	1			1		2	1.8
Chemical Technology	8	3	4	5	4	24	21.6
Plant and Equipment	2	2		1		5	4.5
Electronuclear Research	1	1				2	1.8
Health Physics	2					2	1.8
Isotopes	10	8	4	6	2	30	27.0
Metals and Ceramics			1	1	1	3	2.7
Neutron Physics				2		2	1.8
Operations	8	4			2	14	12.7
Physics	2	1	1			4	3.6
Reactor		2	3		1	6	5.4
Reactor Chemistry	1					1	.9
Solid States					1	1	.9
TOTALS	41	22	16	20	12	111	100.0

8.0 INDUSTRIAL SAFETY

The safety record for 1969 was the second best in the history of the Laboratory. ORNL experienced only two Disabling Injuries during the year. There were fewer Serious Injuries reported this year (1969) than for any year for the past five years. The number of medical treatment cases for 1969 also showed a continuing decline as compared with 1968.

8.1 Accident Analyses

The Disabling Injury Frequency Rate for 1969 was 0.27. The average frequency rate for the previous five years, 1964-1968, was 0.94. The Disabling Injury history of the Laboratory for the five-year period 1965 through 1969 is shown in Table 8.1. The Disabling Injury frequency rates since the inception of Union Carbide as the contractor at ORNL are shown in Table 8.2.

There were 13 divisions which did not have a Serious or Disabling Injury during 1969. There are 16 divisions which have accumulated 1,000,000 or more hours since the last Disabling Injury. The Serious Injury, Disabling Injury, and exposure-hour data for ORNL divisions are shown in Table 8.3.

Table 8.4 includes injury data for the four facilities—ORNL, Paducah, Y-12, and ORGDP. The frequency rates for Disabling Injuries for three of the four Carbide facilities increased in 1969 as compared with 1968. The frequency rates for Serious Injuries also increased at two of the facilities. Serious Injuries at ORNL decreased from 72 in 1968 to 37 in 1969. The frequency rates for Disabling Injuries and Serious Injuries at ORNL for the past five years, 1965-1969, are shown in Table 8.5.

There were 1,175 injuries (includes first aid, Serious Injuries, and Disabling Injuries) reported during 1969. Tables 8.6, 8.7, and 8.8 show injury data according to type of accident, the nature of the injury, and the part of body injured.

8.2 Analyses of Disabling Injuries

Following are brief analyses of the two Disabling Injuries experienced at ORNL in 1969.

Date of Injury - 8/21/69

A machinist was winding emery paper on the end of an aluminum polishing mandrel chucked in a 3/8" variable speed drill motor. The mandrel was a 1/2" aluminum rod, about 11" long, with one end turned down to 3/8" to fit the chuck, and the other end slit lengthwise to hold the paper.

The machinist folded a piece of fine emery paper to five thicknesses, resulting in a piece about 10" by 2". He placed one end of the folded paper in the slit, and with the drill motor lying on the workbench, jogged the switch with his right thumb to rotate the mandrel while guiding the paper with the fingers of his left hand.

He allowed his left index finger to be caught between the rolled paper and the free end, while simultaneously activating the switch. This twisted his finger and jerked the drill from his right hand.

He sustained a dislocation of the first joint of his left index finger. Surgery under general anesthesia was necessary to reduce the dislocation, resulting in one day lost from work.

Date of Injury - 10/21/69

Four men were gathered around a table looking at facility drawings. The table was parallel to a control panel with about a 3-foot aisle between. One of the men was seated at the table in a straight chair with his back to the control panel.

A foreman started from one end of the table to the other in order to see a drawing better. To do so he had to pass between the chair and the panel board, a clearance of 15-18 inches. As he passed behind the chair, he thinks that his pant leg caught on a projection from the panel board, throwing him off stride. His toe caught on a chair leg. He stumbled forward, falling against a metal cabinet and to the concrete floor. He felt pain in his hip immediately and was sent to the dispensary by ambulance.

After the fall, it was observed that the lowest instrument chasis in a rack directly behind the chair had been left projecting about 3 inches from the control panel. The bottom of this chassis was approximately 14 inches from the floor. This could have been the point on which the pant leg caught.

8.3 Safety Awards

A new Safety Incentive Plan was instituted on January 1, 1969, which added Serious Injuries as a factor in award calculation. This and a separation of employees into groups with their award values depending partly on the group's performance gave a more local and personal accent to safety achievement.

a. Installation-Wide Award - When Laboratory employees complete 1,000,000 labor hours without a Disabling Injury, each participant is credited with $1.00.

b. Group Award - When a group works an entire calendar month without a Serious Injury, each member of the group is credited with $0.50. In general, groups consist of individual divisions. Some smaller divisions were combined into single groups when they reported to the same director. Because of its size, Plant and Equipment was subdivided into 22 groups.

Seven 1,000,000-hour injury-free periods were achieved. Groups achieved from a low of nine to a high of 12 months without experiencing a Serious Injury. Thus, annual awards varied from $11.50 to $13.00 ($7.00 installation-wide plus from $4.50 to $6.00 group). Gift certificates were awarded rather than cash or merchandise. Almart Stores of Knoxville was awarded the bid and furnished certificates with a face value of 25 percent above the earned amount.

Table 8.1 Disabling Injury History – ORNL, 1969

	1965	1966	1967	1968	1969
Number of Injuries	18	4	4	1	2
Labor Hours (Millions)	7.7	7.8	8.0	7.8	7.5
Frequency Rate	2.34	0.51	0.50	0.13	0.27
Days Lost or Charged	2816	231	245	60	67
Severity Rate	366	30	31	8	9

Table 8.2

ORNL Disabling Injury Frequency Rates Since Inception of Carbide Contract Compared with Frequency Rates for NSC,[1] AEC and UCC

Year	ORNL	NSC	AEC	UCC
1948	2.42	11.49	5.25	5.52
1949	1.54	10.14	5.35	4.91
1950	1.56	9.30	4.70	4.57
1951	2.09	9.06	3.75	4.61
1952	1.39	8.40	2.70	4.37
1953	1.43	7.44	3.20	3.61
1954	0.79	7.22	2.75	3.02
1955	0.59	6.96	2.10	2.60
1956	0.55	6.38	2.70	2.27
1957	1.05	6.27	1.95	2.41
1958	1.00	6.17	2.20	2.21
1959	1.44	6.47	2.15	2.16
1960	0.94	6.04	1.80	1.92
1961	1.55	5.99	2.05	2.03
1962	1.45	6.19	2.00	2.28
1963	1.55	6.12	1.60	2.10
1964	1.07	6.45	2.05	2.20
1965	2.34	6.53	1.80	2.40
1966	0.64	6.91	1.75	2.57
1967	0.50	7.22	1.55	2.06
1968	0.13	7.35	1.24	2.24
1969	0.27			

[1]National Safety Council (NSC), all industries.

Table 8.3 Injury Record by Division—1969

Division	Medical Treatment Cases	Number of Serious Injuries	Disabling Injuries			Exposure Hours (In Millions)
			Number	Freq.	Sev.	
Analytical Chemistry	24	0	0			0.31
Chemical Technology	35	2	0			0.44
Chemistry	10	1	0			0.19
Director's	9	0	0			0.24
Electronuclear	4	0	0			0.10
Health Physics	24	2	0			0.39
Instr. and Controls	50	3	0			0.58
Mathematics	2	0	0			0.19
Metals and Ceramics	31	0	0			0.59
Neutron Physics	10	0	0			0.14
Physics	7	0	0			0.11
Reactor	3	0	0			0.08
Reactor Chemistry	10	2	0			0.18
Solid State	3	0	0			0.15
General Engineering	6	0	0			0.31
Health	0	0	0			0.06
Isotopes	31	2	0			0.28
Laboratory Protection	7	0	0			0.14
Operations	53	1	1	2.37	157	0.42
Personnel	18	1	0			0.20
Plant and Equipment	777	21	1	0.50	.5	2.01
Technical Information	16	0	0			0.23
Inspection Engineering	9	1	0			0.08
Misc	1	1				0.08
PLANT TOTAL	1140	37	2	0.27	9	7.52

Table 8.4 Four-Plant Tabulation of Injuries—1969

	Labor Hours (Millions)	Disabling				Serious	
		Number of Injuries	Frequency Rate	Days Lost or Charged	Severity Rate	Number of Injuries*	Frequency Rate
ORNL	7.5	2	0.27	67	9	37	4.9
ORGDP	5.1	3	0.60	216	43	38	7.58
Y-12	12.6	4	0.32	349	28	137	10.9
Paducah	2.2	2	0.92	55	25	31	14.28

*Includes the number of Disabling Injuries.

Table 8.5 ORNL Injury Frequency Rates—1965-1969

Year	Serious Injuries	Disabling Injuries
1965	12.60	2.34
1966	11.90	0.64
1967	11.08	0.50
1968	9.3	0.13
1969	4.92	0.27

Table 8.6 Number of Accidents by Types

Type of Accident	Number of Accidents
Struck Against	459
Struck By	313
Slips, Twist	141
Caught In, On, or Between	78
Falls	58
Inhalation, Injection	9
Contact Temp Extremes	10
Electrical	0
Others	107
TOTAL	1175

Table 8.7 Number of Accidents by Nature of Injury

Nature Of Injury	Number of Accidents
Abrasion, Laceration	479
Contusion	219
Strain	137
Eye	36
Burns (Temp)	56
Sprain	27
Burn (Chem)	16
Burn (Flash)	3
Fracture, Dislocation	6
Other	196
TOTAL	1175

Table 8.8 Number of Accidents Relative to Part of Body Injured

Body Area	Percentage	Total Injuries
Eyes	8.7	102
Head	7.2	84
Arms	10.5	123
Shoulder-Chest	2.1	28
Back	9.0	106
Trunk	2.6	31
Hands	11.2	132
Fingers	35.7	419
Legs	7.7	89
Feet	3.7	43
Toes	0.7	8
General	0.9	10
TOTAL	100.0	1175

9.0 LABORATORY ASSAYS

Laboratory Assays Units provide laboratory support to the Health Physics Monitoring Sections. These services include (1) the analysis of body fluids and excreta (bio-assay) for the monitoring of personnel for internal radiation exposure, (2) the radio-chemical analysis of environs samples, (3) counting services for the environs monitoring and radiation survey programs, (4) autoradiography, and (5) whole body counting (in vivo gamma spectrometry).

9.1 Bio-Assay Analysis

The number and types of analyses performed by the Bio-Assay Unit during 1969 are given in Table 9.1. A total of 5,370 analyses were performed which included 5,172 analyses on samples submitted by donors and 198 analyses on standard and blank samples analyzed for control purposes. Approximately 93 percent of the samples were analyzed for either the alpha emitters or strontium.

9.2 Counting Facility

The counting facility processed 188,894 samples during 1969. A tabulation of the number and types of samples counted is presented in Table 9.2. The total number of samples processed was about the same as the previous year.

9.3 Environs Monitoring Sample Analysis

Table 9.3 presents the number and type of environs samples analyzed and the type of analysis performed on each type of sample. A total of 5,169 samples were analyzed during 1969 as compared with 5,274 samples analyzed in 1968. Analysis of environs monitoring samples may range from a single determination to as many as 12 determinations per sample depending upon the radionuclides present. The methods used by the various analytical groups are generally described in the ORNL Master Analytical Manual.

9.4 Autoradiography

There were 1,797 films processed during 1969 in support of radioparticulate studies conducted by the Environs Monitoring Units.[7]

[7] Methods described in ORNL-2601, Radioactive Waste Management at Oak Ridge National Laboratory.

9.5 Whole Body Counter

During the calendar year 1969 the whole body counting program included 1,108 whole body or thorax counts. Nine hundred fifty-two or approximately 86 percent of these counts indicated normal human spectra. There was no case, based on data collected by the IVGS, for which the AEC reportable level for occupational workers (one-half of a permissible body burden averaged over the year) was exceeded.

In addition to the whole body and thorax counts noted above, several counts were made with the wound probe following injuries in contaminated areas. About 200 counting analyses were performed for calibration, radionuclide identification, and development of analytical techniques.

Table 9.1 Bio-Assays Analyses—1969

Analytical Procedure	Number of Analyses	
Urine:		
^{32}P	40	
Trans Pu	606	
Sr	1569	
U	609	
Ce	24	
^{3}H	163	
^{137}Cs	301	
^{239}Pu	1629	
Np	7	
^{60}Co	52	
Ru	26	
Others	56	
Total		5089
Fecal:		
Pu	40	
Sr	3	
U	3	
Trans Pu	38	
Others	6	
Total		90
Standards and blanks		198
TOTAL		5370

Table 9.2 Counting Facility Resume—1969

Types of Samples	Number of Samples			Unit Total	Weekly Average
	Alpha	Beta	Gamma		
Survey Area Samples					
Smears	67,680	69,882		137,562	2,645.4
Air Filters	21,047	21,167		42,214	811.8
Environs Monitoring					
Air Filters	1,711	1,711		3,422	65.8
Gummed Paper		1,660		1,660	31.9
Rain Water		1,566		1,566	30.1
White Oak Lake Effluent	133	1,068		1,201	23.1
Grass			22	22	0.4
Milk			992	992	19.1
Urine			23	23	0.4
Waste Disposal	116	116		232	4.5
GRAND TOTAL	90,687	97,050	1,037	188,894	3,632.6

Table 9.3 Environmental Monitoring Samples—1969

Sample Type	Type of Analyses	Number Samples
1. Monitoring network filters	Gross beta, autoradiogram	1703
2. Gummed paper fallout trays	Gross beta, autoradiogram	1508
3. Rain water	Gross beta	770
4. White Oak Dam effluent	Gross beta, radiochemical, gamma spectrometry	545
5. Clinch River water	Gross beta, radiochemical, gamma spectrometry	20
6. Raw milk	Radiochemical	436
7. Pasture grass	Radiochemical, gamma spectrometry	180
8. Potable water	Radiochemical, gamma spectrometry	7
9. Silt composites	Radiochemical, gamma spectrometry	0
	TOTAL	5169

10.0 HEALTH PHYSICS INSTRUMENTATION

The Health Physics Division shares with the Instrumentation and Controls Division the responsibility for the selection of electronic radiation monitoring instruments used in the ORNL health physics program. Normally, the Health Physics Division is responsible for determining the need for new instrument types and modifications to existing types, for specifying the health physics requirements and for approval of the design. The Health Physics Division is also responsible for calibrating all instruments used in the health physics program and is allocated the funds for maintenance of these instruments. Maintenance is performed or cross-ordered by the Instrumentation and Controls Division.

Non-electronic personnel monitoring devices are designed, tested, calibrated and maintained by Health Physics Division personnel.

10.1 Instrument Inventory

The electronic instruments used in the health physics program are divided, for convenience in servicing and calibrating, into two classes: the first class includes battery-powered portable instruments; the second class includes the stationary instruments that are AC powered. Portable instruments are assigned and issued to the Radiation and Safety Surveys Units. Stationary instruments are the property of the ORNL division which has the monitoring responsibility in the area in which the instrument is located. Table 10.1 lists portable instruments assigned at the end of 1969; Table 10.2 lists stationary instruments at the X-10 site in use at the end of 1969. There were net increases in 1969 of 60 portable instruments and 23 stationary instruments.

During 1969, 500 new pocket meters, 318 new fiber dosimeters (200 mR range) and 33 personal radiation monitors (PRM) were issued by ORNL Stores. The pocket meters issued were replacements for instruments which had been lost or damaged.

Inventory and Service Summaries for health physics instruments are prepared on an IBM 360. These computer programmed reports enable the Instruments Group to maintain a current inventory on most health physics instrument requirements.

The allocation of stationary health physics monitoring instruments at the X-10 site by division is shown in Table 10.3.

10.2 Calibration Facility

The Health Physics Division maintains a calibration facility for the calibration and maintenance of portable radiation instruments and personnel metering devices. The facility is equipped with calibration sources, remote control devices, and shop space for the use of Instrumentation and Controls Division maintenance personnel. Health Physics personnel assign, arrange for maintenance of, calibrate, provide delivery services for, and maintain inventory and servicing data of all portable health physics instruments.

Portable instruments should be serviced (1) whenever repairs are needed, (2) at least once each two months for those which have replacement-type batteries, and (3) at least once each three months for those instruments which have "permanent" (rechargeable) batteries. The number of calibrations of portable instruments for 1969 is shown in Table 10.4.

Stationary instruments are calibrated by Calibrations Group personnel or by Radiation and Safety Surveys personnel who use sources which are designed, standardized, and provided by the Calibrations Group.

Table 10.1 Portable Instrument Inventory—1969

Instrument Type	Instruments Added 1969	Instruments Retired 1969	Assigned Inventory Jan. 1, 1970
GM Survey Meter	24	0	480
Cutie Pie	14	0	464
Alpha Survey Meter	22	0	260
Neutron Survey Meter	0	0	99
Miscellaneous	0	0	40
TOTAL	60	0	1343

Table 10.2 Inventory of Facility Radiation Monitoring Instruments for the Year—1969

Instrument Type	Installed During 1969	Retired During 1969	Total Jan. 1, 1970
Air Monitor, Alpha	7	0	100
Air Monitor, Beta	2	0	182
Hand-Foot Monitor	2	1	33
Lab Monitor, Alpha	5	0	152
Lab Monitor, Beta	2	0	213
Monitron	3	3	238
Other	6	0	251
TOTAL	27	4	1170

Table 10.3 Health Physics Facility Monitoring Instruments
Divisional Allocation at X-10 Site—1969

ORNL Division	α Air Monitor	β Air Monitor	α Lab Monitor	β Lab Monitor	Monitron	Other	Total
Analytical Chemistry	6	14	13	18	15	6	72
Chemical Technology	49	49	53	35	38	37	261
Chemistry	9	9	19	24	19	8	88
Metals and Ceramics	12	16	16	17	11	16	88
Isotopes	15	30	22	41	53	18	179
Operations	1	41	2	19	53	14	130
All Others	8	23	27	59	49	186	352
TOTAL	100	182	152	213	238	285	1170

Table 10.4 Calibrations Resume—1969

		1968	1969
A.	Portable Instruments Calibrated		
	1. Beta-Gamma	3,755	3,778
	2. Neutron	229	350
	3. Alpha	1,100	943
	4. Pocket Chambers and Dosimeters	1,779	1,833
B.	Films Calibrated		
	1. Beta-Gamma	1,598	2,344
	2. Neutron	20	16

11.0 PUBLICATIONS AND PAPERS

D. M. Davis, Applied Health Physics and Safety Annual Report for 1968, ORNL-4423, July, 1969.

E. D. Gupton, "Alpha Air Monitor for ^{239}Pu", Note Published in Health Physics, Vol. 16, No. 6, p. 808-810, June, 1969.

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G. D. Kerr and J. S. Cheka, "A Lithium-7 Phosphate Glass Detector for Exposure Measurements in Mixed Neutron Gamma-Ray Radiation Fields", Note Published in Health Physics, Vol. 16, No. 2, p. 231-232, February, 1969.

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